NON-PUBLIC?: N
ACCESSION #: 8804270321
LICENSEE EVENT REPORT (LER)

FACILITY NAME: Plant Hatch, Unit 2 PAGE: 1 of 10

DOCKET NUMBER: 05000366

TITLE: Calibration Procedural Deficiency For Feedwater Controller
Causes Low Water Level Scram
EVENT DATE: 03/21/88 LER #: 88-008-00 REPORT DATE: 04/20/88

OPERATING MODE: 1 POWER LEVEL: 018

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR SECTION
50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:
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SUPPLEMENTAL REPORT EXPECTED: No

ABSTRACT: On 3/21/88 at approximately 1644 CST, Unit 2 was in the run mode of operation at an approximate power level of 440 MWt (approximately 18% of rated thermal power). In the process of transferring the Feedwater Control System (FCS EIIS Code SJ) from startup control to single element control, fluctuations in reactor vessel water level were experienced. Operations personnel manually scrammed the reactor due to decreasing level and also received a Primary Containment Isolation System (PCIS EIIS Code JM) valve Group 2 isolation. Level was restored using the reactor feed pumps and manually initiating Reactor Core Isolation Cooling (RCIC EIIS Code BN).

The immediate cause for the FCS instability was improper settings on the master control loop controller (2C32-K636). The root cause for these improper settings was determined to be a deficient calibration procedure.

Corrective actions for this event included correcting the settings on the controller and scheduling development of a more comprehensive calibration procedure. As an enhancement, a comparable procedure will also be developed for Unit 1.

(End of Abstract)

A. REQUIREMENT FOR REPORT

This report is required per 10 CFR 50.73 (a)(2)(iv), because unplanned actuations of the Reactor Protection System (RPS EIIS Code JC) and an Engineered Safety Feature (ESF) occurred. Specifically, RPS was manually activated due to a decreasing reactor water level. The ESF which actuated was the Primary Containment Isolation System (PCIS EIIS Code JM) valve Group 2.

B. UNIT(s) STATUS AT TIME OF EVENT

1. Power Level/Operating Mode

Unit 2 was starting up following the unit's seventh refueling outage with an approximate power level of 440 MWt (approximately 18 percent of rated thermal power). The reactor mode switch was in the run position.

2. Inoperable Equipment

There was no inoperable equipment that contributed to this event.

C. DESCRIPTION OF EVENT

1. Event

On 3/21/88 at approximately 1644 CST, the Feedwater Control System (FCS EIIS Code SJ) was in the startup level control mode of operation. In this condition, the startup control station (2C32-R619) automatically directs the opening and closing of a startup level control valve (2N21-F111) to maintain reactor water level, using one Reactor Feed Pump (RFP EIIS Code SJ) in manual to supply required flow. The startup level control valve was approximately 95% open.

At this point, the FCS must be transferred from startup control to single element control, per procedure 34SO-N21-007-2S (Condensate and Feedwater System), in order for further increases in reactor power to be made. In single element control, the RFP speed is automatically adjusted by a master controller to maintain reactor water level.

At 1645 CST, to perform this transfer from startup level to single element master control, licensed Operations personnel placed the master control station (2C32-R600) in the automatic mode. In addition, Operations personnel placed the startup control station (2C32-R619) in manual control. They then began manually opening the startup level control bypass valve (2N21-F110).

The bypass valve (2N21-F110) is a 24-inch valve in parallel with the 12-inch startup level control valve. The reactor water level had started to increase with the opening of the bypass valve as expected. Operations personnel closed the startup level control valve approximately 50% to reduce the expected reactor water level fluctuation that normally occurs when the bypass valve is opened. As the level started to fluctuate, the operator adjusted the setpoint tape on the master controller (2C32-R600) to try to lessen the reactor water oscillations.

Operations personnel noted that the 2A RFP speed was initially sluggish in response to the setpoint tape movement. In addition, they noted that the oscillation in reactor water level was greater than anticipated. As the fluctuations in the reactor water level became more pronounced, especially on the downward side, a "Low Condensate Booster Pump Suction Pressure" alarm was received.

At 1646 CST, the 2A RFP increased speed and then tripped on low suction pressure. Operations personnel tried without success to restart the 2A RFP from the main control room. An operator was dispatched to locally restart the pump. The reactor water level continued to decrease.

Operations personnel started the 2B RFP, but it could not get up to speed fast enough. Reactor water level was approximately 26 inches above instrument zero and still decreasing.

At 1647 CST, a low water scram on RPS channel A was received. Operations personnel, recognizing the impending automatic low water level trip, inserted a manual scram. The reactor water level was approximately 15 inches above instrument zero. Shortly thereafter, a PCIS valve Group 2 isolation signal was received due to low reactor water level.

At approximately 1648 CST, the 2A RFP was locally restarted. Reactor water level was now approximately 29 inches below instrument zero. To help restore level and in anticipation of its automatic initiation
Reactor Core
Isolation Cooling (RCIC EIIS Code BN) was manually started by Operations personnel. The lowest reactor water level reached during the event was approximately 30 inches below instrument zero which is 134 inches above the Top of Active Fuel (TAF).

At approximately 1651 CST, the 2A RFP was shutdown as the reactor water level was recovering. At approximately 1658 CST, the RCIC system was shut off as reactor water level had stabilized in the normal operating range at approximately 34 inches above instrument zero.

During this event, reactor pressure varied between approximately 920 and 260 psig. The low power level prior to the scram and the injection of cold water from RCIC contributed to the pressure decrease following the scram. At approximately 1707 CST, the Main Steam Isolation Valves (MSIVs EIIS Code JM) were manually closed by Operations personnel to control the reactor vessel cooldown rate.

At 1817 CST, the NRC was notified of the actuation of the RPS and the PCIS valve Group 2 isolation due to low reactor water level, per 10 CFR 50.72 reporting requirements.

2. Dates/Times

Date Time (CST) Description

3/21/88 1644 FCS was in the startup level control mode of operation. Startup level control valve was approximately 95% open.

1645 Operations personnel began the transfer from startup control to single element control per procedure 34SO-N21-007-2S.

Date Time (CST) Description

3/21/88 1645 Master control station (2C32-R600) was placed in automatic, startup control station (2C32-R619) was placed in manual, and startup level control bypass valve (2N21-F110) was opened.

Reactor level began to fluctuate more than expected. Closing the startup level control valve (2N21-F111) approximately 50% and increasing the speed of the 2A RFP did not prevent level from decreasing.

1646 2A RFP tripped on low suction pressure and could not be restarted from the control room. Operator was sent to locally restart the RFP.

2B RFP started but level continued to decrease.

1647 Received a low reactor water level half scram on RPS channel A. Operations personnel manually scrammed the reactor with reactor water level at approximately 15 inches above instrument zero.

Received a PCIS valve Group 2 valve isolation.

1648 2A RFP locally restarted. With reactor water level now approximately 29 inches below instrument zero, Operations personnel manually started RCIC.

1651 2A RFP manually shutdown as reactor water level recovered.

Date Time (CST) Description

1658 RCIC shutdown as reactor water level
stabilized in normal operating range.

1707 Operations personnel closed MSIVs to
control cooldown rate of reactor
vessel.

1817 The NRC was notified of the actuation
of RPS and the Group 2 isolation per
10 CFR 50.72 reporting requirements.

3. Other Systems Affected

No safety systems, other than the RPS and PCIS valve Group 2, were affected by this event. These systems have no secondary functions.

4. Method of Discovery

The FCS instability was discovered by Operations personnel during the transfer from startup control to single element control per procedure 34SO-N21-007-2S (Condensate and Feedwater System).

5. Operator Actions

Operations personnel performed the following actions:

1. When attempts to control the FCS instability failed, manually inserted the reactor scram when reactor water level approached the low level scram setpoint.

2. Started the 2A and 2B RFPs and RCIC to successfully stabilize water level and closed the MSIVs to control vessel cooldown rate.

3. Initiated an Event Review Team (ERT) investigation per plant administrative guideline AG-MGR-31-0787N.

Nuclear Safety and Compliance (NSC) personnel performed the following action:

1. Participated in the event investigation and
reported the event per 10 CFR 50.73 requirements.

6. Auto/Manual Safety System Response

A manual RPS actuation occurred and scrammed the reactor. Also, a PCIS valve Group 2 isolation occurred and all Group 2 valves closed as required per design on low reactor water level.

D. CAUSE OF EVENT

1. Immediate Cause

The immediate cause for the FCS instability was determined to be improper settings on the master controller module (2C32-K636). As the result of a review of the maintenance history of this component, it appears the settings were inadvertently changed during a repair performed under Maintenance Work Order (MWO) 2-87-4727 (completed on 2/10/88).

The failure of the 2A RFP to be restarted from the control room was a contributing factor in the event. The remote reset switch of the 2A RFP in the main control room did not function properly. This was caused by the incorrect wiring of the reset limit switch (LS-7). The wiring was set up for "normally closed contacts" in lieu of "normally open contacts" as required by design drawings. The maintenance history of the switch over Unit 2's seventh refueling outage was reviewed, and it could not be determined when the switch was wired incorrectly.

2. Root/Intermediate Cause

The root cause for the FCS instability, due to the improper settings on the master controller module (2C32-K636), was determined to be a procedural deficiency.

To perform the maintenance required under MWO 2-87-4727, all three of the FCS master control loop instruments were removed from service. The master control loop is composed of three distinct instruments: 1) the setpoint manual/automatic (M/A) station (2C32-R600), 2) the control amplifier (2C32-K637), and 3) the controller (2C32-K636).

The maintenance performed on the three instruments consisted of the following. An automatic switch was replaced in 2C32-R600. In addition, a capacitor was replaced on 2C32-K637.

Following completion of this maintenance, a calibration of the FCS master control loop instruments was performed per procedures 57CP-CAL-048-02 (General Electric 543-03 Controller Amplifier) and 57CP-CAL-044-2 (GE Type 547-01 Self Synchronizing M/A Transfer Station). Although procedure 57CP-CAL-048-02 is specifically for calibration of the amplifier (2C32-K637), it requires that all three components in the control loop be removed from service to calibrate 2C32-K637. Procedure 57CP-CAL-044-2 only covers the setpoint M/A station (2C32-R600) and was used to support the work performed under procedure 57CP-CAL-048-02.

Maintenance records document the calibration as being checked after work on the MWO was complete. The exact period when the settings were incorrectly changed on the controller (2C32-K636) from the correct values existing prior to the outage cannot be identified.

Procedure 57CP-CAL-048-02 was identified as being deficient. The procedure does not cover the removal of the three master control loop instruments from service, nor does it require recording the as-found and as-left conditions (settings) for 2C32-K636 and 2C32-K637. Therefore, the incorrect settings that occurred on 2C32-K636 are attributed to a task not being properly covered by the applicable procedure.

E. ANALYSIS OF EVENT

The RPS provides timely protection against the onset and consequences of conditions that could threaten the integrities of the fuel barriers and the nuclear system process barrier.

A low water level in the reactor vessel indicates that the reactor is potentially in danger of being inadequately cooled. Should reactor water level decrease too far, fuel damage could result. A reactor scram, initiated by a low water level condition, protects the fuel by reducing the fission heat generation within the core.

In this event, the decrease in the vessel level was correctly sensed by the RPS, as indicated by the half scram received on RPS channel A. Operations personnel anticipated the full scram on low water level and manually scrammed the reactor. The RPS functioned per design. Operations personnel restored reactor water level by using the RFPs and RCIC; therefore, High Pressure Coolant Injection (HPCI EIIS Code BJ) was not required to operate.

These prompt corrective actions rapidly terminated power operations (and energy generation) and restored monitored plant parameters (such as reactor water level) to their nominal values.

Based on the above information, it is concluded that this event had no adverse impact on nuclear safety. While this event occurred at a low reactor power level, the above analysis is applicable to all power levels.

F. CORRECTIVE ACTIONS

The corrective actions for this event included:

1. Returning the proportional and reset settings on the master control module (2C32-K636) to their pre-outage values per MWO 2-88-1696.

2. Developing a more comprehensive procedure for Unit 2 to calibrate the FCS master control loop. The procedure will record the as-found settings and as-left settings for the 2C32-K636 and 2C32-K637 instruments. It should be noted that the deficient procedure, 57CP-CAL-048-02, had not been through the Procedure Upgrade Program, a long term program to upgrade procedure format and correct any possible procedure deficiencies. As an enhancement, a comparable procedure will be developed for Unit 1. The procedures will be approved for validation by 9/1/88, with an estimated completion date of 1/17/89.

3. Correcting the wiring on limit switch (LS-7) for the 2A RFP under MWO 2-87-3816.

G. ADDITIONAL INFORMATION

1. FAILED COMPONENT(s) IDENTIFICATION

There was no component failure experienced in this event.

2. PREVIOUS SIMILAR EVENTS

There have been three similar events to the one described in this LER. They were reported in the following LERs: 50-366/1986-012 (dated 6/28/86), 50-321/1986-030 (dated 8/2/86), and 50-366/1987-008 (dated 4/22/87).

These LERs describe events where either feedwater flow was lost or feedwater oscillations occurred which resulted in reactor scrams.

These events were caused respectively by failure of flow control valve 2N21-F007, setpoint error on the minimum flow bypass valves, and a hard electrical ground condition in a condensate pump motor. Corrective actions for these evnts included: 1) equipment repair, 2) correcting deficient procedures, 3) reviewing maintenance histories, and 4) scheduling future maintenance.

The corrective actions for these events would not have prevented the event described by LER 50-366/1988-008 because the causes of the events were different as noted above.

ATTACHMENT # 1 TO ANO # 8804270321 PAGE: 1 of 2

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April 20, 1988

U. S. Nuclear Regulatory Commission
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Washington, D. C. 20555

PLANT HATCH - UNIT 2
NRC DOCKET 50-366
OPERATING LICENSE NPF-5
LICENSEE EVENT REPORT
CALIBRATION PROCEDURAL DEFICIENCY FOR
FEEDWATER CONTROLLER CAUSES LOW WATER LEVEL SCRAM

Gentlemen:

In accordance with the requirements of 10 CFR 50.73(a)(2)(iv), Georgia Power Company is submitting the enclosed Licensee Event Report (LER) concerning the unanticipated actuation of some Engineered Safety Features (ESFs). The event occurred at Plant Hatch - Unit 2.

Sincerely,
/s/ L. T. Gucwa
L. T. Gucwa

CLT/1c
Enclosure: LER 50-366/1988-008

c: (see next page)

Georgia Power

U. S. Nuclear Regulatory Commission
April 20, 1988
Page Two

c: Georgia Power Company
Mr. J. T. Beckham, Jr., Vice President - Plant Hatch
GO-NORMS

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